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Sandia National Laboratories Medical Isotope Reactor Concept

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Abstract

This report describes the Sandia National Laboratories Medical Isotope Reactor and hot cell facility concepts. The reactor proposed is designed to be capable of producing 100% of the U.S. demand for the medical isotope ^{99}Mo. The concept is novel in that the fuel for the reactor and the targets for the ^{99}Mo production are the same. There is no driver core required. The fuel pins that are in the reactor core are processed on a 7 to 21 day irradiation cycle. The fuel is low enriched uranium oxide enriched to less than 20% ^{235}U. The fuel pins are approximately 1 cm in diameter and 30 to 40 cm in height, clad with Zircaloy (zirconium alloy). Approximately 90 to 150 fuel pins are arranged in the core in a water pool ~30 ft deep. The reactor power level is 1 to 2 MW. The reactor concept is a simple design that is passively safe and maintains negative reactivity coefficients. The total radionuclide inventory in the reactor core is minimized since the fuel/target pins are removed and processed after 7 to 21 days. The fuel fabrication, reactor design and operation, and ^{99}Mo production processing use well-developed technologies that minimize the technological and licensing risks. There are no impediments that prevent this type of reactor, along with its collocated hot cell facility, from being designed, fabricated, and licensed today.

Acknowledgement

The authors would like to thank Milt Vernon and Mike Gregson for their input and many discussions on the conceptual ideas of a Medical Isotope Reactor; Paul Helmick and Derek Aaronscooke for their contribution to the facility layout; and Paul Raglin, John Kelly, Gary Rochau, Dave Wheeler, and Andrew Orrell for their continued support of this work.

CONTENTS

FIGURES

EXECUTIVE SUMMARY

Need

The history and current status of ^{99}Mo supply to the U.S. medical system is well-documented and will not be repeated in detail here. Suffice it to say that the current fleet of nuclear reactors throughout the world that produce ^{99}Mo from the fission process are aging and cannot meet the world demand. New reactors built in Canada have never come online due to safety issues. Highly enriched uranium (93% ^{235}U) is used for the target fuel material throughout the world, which presents a proliferation concern. Also, there is no domestic (U.S.) supplier for the raw ^{99}Mo product. Hence there is a great need to develop a simple approach for ^{99}Mo production that can meet the U.S. demand (USD) using low enriched fuel, with the goal of making the system safe, cost effective, and readily achievable, using existing and proven technologies.

Requirements

The only viable path to providing an adequate supply of ^{99}Mo for the required specific activity is through fission. To satisfy the current USD, ~6,000 six-day curies (Ci) must be produced per week. The world demand, apart from the USD, is approximately the same as that of the USD. The U.S. and world demand is expected to grow a few percent each year. Production of 6,000 six-day Ci per week requires at least 1.1 MW of continuous target fission power, assuming two post-irradiation days for processing and shipping. Irradiation times must be short (7 to 21 days) to control quality in purity levels and specific activity.

SNL Prior Experience

Sandia National Laboratories (SNL) proposes to draw upon the experience gathered from its Department of Energy (DOE)-sponsored ^{99}Mo medical isotope program in the 1990's. That experience includes reactor modifications for ^{99}Mo target irradiation, hot cell facility and process equipment design and fabrication for ^{99}Mo processing, transfer cask design and fabrication, reactor and hot cell facility safety analysis reports, the NEPA process leading to a Record of Decision, target irradiation, ^{99}Mo extraction, QA inspection, and product and waste packaging and shipping. The previous work at SNL was based on the Cintichem process, which is a precipitation process for separation of ^{99}Mo from oxide fuels.

Proposed Concept

The most direct approach to meet the domestic ^{99}Mo demand is to use a small, passively safe reactor based on proven technology and low enriched uranium (LEU) fuel, dedicated solely to the production of ^{99}Mo. The concept presented here is based on using the reactor fuel as the targets. With this approach there is no separate driver core and target region. The quantity of ^{99}Mo produced is directly proportional to the power attainable in the fuel pins. Using this approach, ^{99}Mo can be produced in a cost-effective manner with little or no additional development of new techniques or processes and without uncertain licensing issues. There are no impediments in keeping this concept, along with its collocated hot cell facility, from being designed, fabricated, and licensed today.

The SNL Medical Isotope Reactor concept consists of a small 1 to 2 MW open pool-type reactor. The reactor active core region is approximately 30 cm in diameter and 30 to 40 cm in fuel height. It contains approximately 90 to 150 cylindrical, 1-cm diameter, LEU oxide fuel pins clad with

Zircaloy (zirconium alloy). The pins serve as the ^{99}Mo targets and, if operated with an average pin power of 8 to 10 kW, can meet the required ~1.1 MW of target power for 100% of the USD. The fuel pins are cooled by natural-circulating water through the core. If operated at a higher power level or with a greater number of fuel pins, the production would be greater than the USD.

The technology used in the concept is proven, based on current and past research reactors and commercial power systems. No new technologies need to be developed. Fuel/target pin fabrication is based on existing light water reactor (LWR) fuel fabrication processes. The safety/control system is simple and can be purchased off the shelf from research reactor providers such as those used for TRIGA reactor systems.

An adjacent hot cell facility would be connected to the reactor pool by a water channel to facilitate remote transfer of irradiated targets for processing. Processing would be accomplished using well-known and successful oxide dissolution and separation processes. Processing, QA, packaging, and shipping would follow the well-developed Cintichem process. A storage area would be provided to store waste for ~6 months to 1 year prior to disposal. Reprocessing of the fuel remains an open option, depending on the economics of disposal versus recovery of the uranium.

The reactor power of 1 to 2 MW is typical of small university systems, and the removal of targets after a 7 to 21 day irradiation cycle precludes the build-up of a significant radionuclide inventory in the reactor core. The fuel pins are adequately cooled by using natural circulation. No emergency core cooling capability or backup power supply is needed for the reactor. The pool water is cooled by a secondary heat rejection system. The pool water, in addition to providing cooling, serves as shielding for personnel and will retain radionuclides in the event of a fuel pin leak or cladding failure.

The reactor is similar to university reactors in power, hardware, and safety/control systems. Thus, the current Nuclear Regulatory Commission (NRC) experience with university-type reactors is directly applicable and the licensing process should be straightforward and relatively uncomplicated.

Summary

The USD for ^{99}Mo can be met with existing and well-proven fuel fabrication processes, research reactor technologies, and ^{99}Mo processing procedures. The SNL Medical Isotope Reactor concept represents the lowest cost option, the shortest time path option, and the lowest regulatory and business risk option to meet and/or exceed the USD with one facility. There are no impediments that prevent this reactor concept, along with its collocated hot cell facility, from being designed, fabricated, and licensed today.

A provisional patent for the SNL Medical Isotope Reactor concept was filed on 11/9/2009.

1 INTRODUCTION

The history and current status of ^{99}Mo supply to the U.S. medical system is well-documented and will not be repeated in detail here. Abundant information can be attained through the Internet and References 1 to 3. Virtually all of the ^{99}Mo produced in the world today is generated by way of the fission process. Driver reactor cores operating at tens to hundreds of megawatts produce a high neutron flux in an irradiation region where targets, using highly enriched uranium (HEU) as fuel, are irradiated continuously for 7 to 14 days. After irradiation, the targets are transferred and processed at a hot cell facility to separate the ^{99}Mo product isotope. The waste stream, including the uranium, is then stored for decay and disposed of at a later date. Currently, target fuel is not reprocessed. Several problems exist worldwide with the current production scenario. These include the following:

- There are not enough worldwide suppliers of ^{99}Mo to keep up with the growth of the market and planned and unplanned outages. The MAPLE reactors in Canada, which were to produce significant quantities of ^{99}Mo, have never come online due to safety issues in their design.
- There is no domestic U.S. supplier for ^{99}Mo. The last U.S. supplier was Cintichem, Inc., which ceased operations in the late 1980's.
- HEU fuel is used in all current major production applications, adding to the proliferation risks and uncertainty in future supply.
- The current high-flux reactors used for production were initially built for high-flux applications, not primarily or solely for ^{99}Mo production. The ^{99}Mo production effort has always been an add-on application to these types of reactors. No reactor has ever been built with the sole purpose of optimizing for ^{99}Mo production, using cost effectiveness as the driving constraint.
- The current high-flux reactors used for production are owned by government agencies or universities and are not operated for the sole purpose of producing ^{99}Mo. This causes two problems. First, the true cost for the production of ^{99}Mo becomes difficult to ascertain since these reactors are subsidized in the costs of operation, maintenance, and refueling. Second, the demands of other customers, in addition to the production of ^{99}Mo, cause conflicts in scheduling.
- The current high-flux reactors used for production are expensive to operate, maintain, and refuel. Profit margins are difficult to maintain to the degree necessary to keep the reactor facility operational.
- Many of the current high-flux reactors used for production are 40 to 50 years old and are suffering problems with their age. Outages are becoming more frequent and severe. Tank leakage has forced the current outage of the NRU reactor in Canada, which supplied 60% USD. The HFR reactor in the Netherlands, which supplied 40% USD and was major supplier to Europe, is now down for months of repair due to tank leakage.

There is a need to develop a simple approach to ^{99}Mo production that can meet the USD using LEU fuel. The goal should be to build a facility with the capacity to produce the USD for ^{99}Mo while being passively safe, cost effective, and readily achievable using existing and proven technologies. The development of a reactor/hot cell facility to solely produce ^{99}Mo based on

LEU fuel, cost effectiveness, and simplicity should be the objective. The reactor concept should have a very high technology readiness level to allow design, licensing, and construction to begin within a short period of time.

With this goal in mind, the Sandia SNL Medical Isotope Reactor concept was developed (Ref. 4-6). The most important feature of the concept is that the reactor fuel pins are the same as the targets and use LEU fuel. There is no separate driver core for the reactor. Depending on the production level required, fuel pins can be removed and processed on a daily basis. The removed fuel pins are replaced in the core with fresh fuel pins and the reactor is restarted and operated at full power until the next removal cycle. One or several fuel pins can be removed at each cycle. The reactor can also be shut down for fuel pins to be removed and then restarted many times during the day, if desired. The main features of this concept are as follows:

- Reactor fuel pins are the same as the targets. There is no driver core. The fuel pins achieve criticality and the reactor power is controlled using control rods and a control system similar to low-power research reactors.
- Simple and robust design of the reactor with passive safety features.
  - Low power level, 1 to 2 MW. Relatively low fission product inventory.
  - Small size and number of fuel pins. The reactor core size is only ~30 cm in diameter and 30 to 40 cm in fuel height. The number of fuel pins required is only 90 to 150.
  - Negative reactivity coefficients. Reactor shuts down as power level increases.
  - Natural-convection cooling of the core. Tank-type design with the core in ~30 ft of water. Pool is cooled using a secondary heat exchanger and a small cooling tower.
  - No emergency core cooling is required. No standby or backup electrical power is required. Loss of power shuts down the reactor by dropping electromagnetically-coupled control rods. Decay power is removed from the core by natural-convection cooling. The pool maintains a significant heat capacity for passive cooling.
- Fuel pins use LEU fuel and well-developed fabrication processes. The fuel pins are ~1 cm in diameter. The fuel form is oxide fuel-stacked pellets in Zircaloy cladding. The fuel height in each pin is 30 to 40 cm.
- Control system is simple and uses well-developed research reactor technologies.
- Fuel pins are transferred to a collocated hot cell facility and processed using the well-developed Cintichem process or other separation techniques.
- Production capacity is greater than 100% of USD for ^{99}Mo.
- Licensing approach through NRC should be straightforward.
- Cost-effective approach. The concept can be shown to be cost effective and profitable using 20% enriched fuel pins at a production level of 20% of the USD for ^{99}Mo.

Using this approach, ^{99}Mo can be produced in a cost-effective manner with little or no additional development of new techniques and processes and without uncertain licensing issues. There are no impediments that prevent this reactor, along with its collocated hot cell facility, from being designed, fabricated, and licensed today.

2 SNL'S EXPERTISE RELATED TO MEDICAL ISOTOPE PRODUCTION

Sandia National Laboratories proposes to draw upon the experience gained from its DOE-sponsored ^{99}Mo medical isotope program in the 1990's. That experience includes reactor modifications for ^{99}Mo target irradiation, hot cell facility and process equipment design and fabrication for ^{99}Mo processing, transfer cask design and fabrication, reactor and hot cell facility Safety Analysis Reports, the NEPA process leading to a Record of Decision (ROD), target irradiation, ^{99}Mo extraction, quality assurance (QA) and quality control (QC) processes, and product and waste packaging and shipping. The work was based on the Cintichem process in which Cintichem targets, internally coated with HEU oxide fuel, were to be irradiated in the central region of the SNL Annular Core Research Reactor (ACRR). Test irradiations, processing, and product shipment were performed to demonstrate the process feasibility and product purity. SNL remains the last U.S. site where the Cintichem process was exercised on a full-scale target irradiated for a full-production cycle of seven days, with the product shipped to a radio-pharmaceutical company for evaluation. SNL still maintains the procedures required to use the Cintichem production process. Publications from this previous work are provided in References 7 to 21.

Although SNL has not recently had a role in medical isotope production, it has maintained its proficiency in small reactor designs, e.g., space reactors, power conversion cycles, reactor in-pile experimentation, and operation of several nuclear facilities. The nuclear facilities operated at SNL in Technical Area V (TA-V) include three reactors: the ACRR, Sandia Pulse Reactor (SPR), and Critical Assembly (CA). The Gamma Irradiation Facility (GIF) is also operated at TA-V. The SNL Hot Cell Facility (HCF), which was partially modified but not completed as part of the medical isotope program, remains in standby status. Most of the key members of the DOE-sponsored ^{99}Mo medical isotope program still remain employed at SNL TA-V and are involved in experimentation and advanced nuclear concepts development. The use of their knowledge base and lessons learned has contributed greatly in the development of the ^{99}Mo reactor concept presented in this document.

SNL has extensive safety analysis documentation experience through the operation of the TA-V nuclear facilities. SNL gained considerable experience in the unique aspects of ^{99}Mo production, reactor and hot cell design and modification, and safety basis documentation as part of the DOE-sponsored ^{99}Mo medical isotope program. The SNL effort also included participation in the NEPA process which led to a Record of Decision.

The previous medical isotope production program at SNL was based on using Cintichem-type targets. These targets, produced at Los Alamos National Laboratory, were ~1-inch diameter stainless steel tubes, ~0.030 inches in thickness, and coated internally with ~30 grams of HEU (93% enriched) oxide fuel. The targets were sealed with end caps and leak-tested as part of the QC testing. The plan was to irradiate 19 to 37 of these targets in the central region of the ACRR on a seven-day irradiation cycle in order to meet or exceed the USD for ^{99}Mo. The project was halted in 1999 with the assurance from Canada that the MAPLE reactors would soon be operating and fulfill the world's needs for ^{99}Mo.

Cintichem-type targets will not be used for the SNL Medical Isotope Reactor concept presented in this document for a number of reasons. The Cintichem process for separating the ^{99}Mo from the fuel and fission products is still considered a viable processing method for UO_2 fuel. Cintichem targets, however, were originally designed for use with HEU fuel. The objective was to minimize the quantity of HEU fuel but maximize the heat transfer to allow for a high target power. This would allow for a high ^{99}Mo product yield per mass of ^{235}U. The HEU Cintichem-type targets were very expensive to manufacture and store due to the difficult and time-consuming coating process and the security issues associated with the HEU fuel.

The Cintichem-type target design is not readily applicable to LEU fuel since five times as much UO_2 must be used for 20% enriched LEU fuel. The coating process is an art that is not readily applicable to thicker coatings. In order to use LEU fuel, foils or packed beds must be considered. These concepts also have issues with manufacturability, processing, and cost.

In order to use LEU fuel in a ^{99}Mo target, a better approach is to use the fuel in a pellet form and a more conventional pin type element with cladding. Using this approach, the heat flux and the fuel pellet centerline temperature will limit the maximum power in the target. The Cintichem chemical separation process can still be used on the oxide pellet fuel form. Although the ^{99}Mo yield per mass of ^{235}U may be less than a Cintichem-type target, the pellet-target configuration will be much more cost effective to manufacture, inspect, and process compared to a coating, foil, or packed bed. Pellet and pin fabrication would use proven LWR fuel technology. The pellet fabrication, pin assembly, and QC inspection would be facilitated through a commercial nuclear fuels vendor.

3 REACTOR DESIGN CONCEPT

The proposed SNL Medical Isotope Reactor concept is an open pool-type reactor, which offers the greatest flexibility in target inventory management (shielded retrieval, replacement, and transfer) and provides inherent passive safety features that lower the safety and operation interruption risk. Figure 1 shows the conceptual layout of the reactor pool, transfer pool and hot cell facility. The reactor is submerged in a water pool sufficiently deep (~30 ft) to afford personnel shielding and of sufficient diameter to assure ample water for passive natural-circulation cooling. A pool cooling loop to a plate-type heat exchanger and secondary loop with an external cooling tower is necessary to maintain pool water temperature to ~40 C.

The main design feature of this reactor concept is that the fuel pins are the targets for ^{99}Mo production and use LEU fuel. Since the fuel pins and the targets are one and the same, they will be referred to throughout the report as fuel/target pins, fuel pins, or targets. The fuel/target pins are 1) irradiated in the reactor grid, 2) transferred under water through the transfer pool channel to the hot cell facility, and 3) chemically processed to extract the ^{99}Mo in the hot cell. The hot cell is also used for QC sampling, product packaging, and waste handling, packaging, and storage until subsequent disposal.

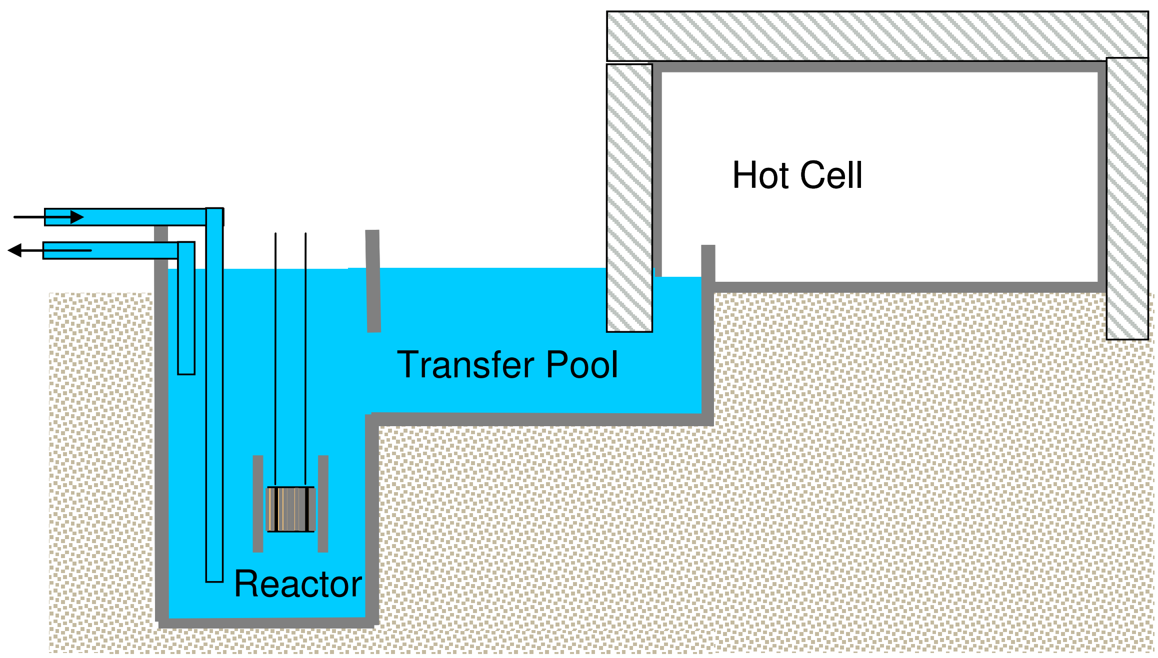

Figure 1. Conceptual ^{99}Mo Facility Layout.

3.1 Fuel Pin Design and Reactor Configuration

A variety of fuel options are available for use in the reactor, but UO_2 fuel is selected as the baseline case since it is readily fabricated based on existing LWR fuel fabrication. Oxide fuel is also compatible with the Cintichem ^{99}Mo extraction technology. UO_2 fuel represents the lowest cost, highest technical feasibility, and lowest programmatic risk option.

The baseline case reactor consists of 90 to 150 LEU (20% enriched) UO_2 fuel pins arranged in a triangular configuration with a pitch (distance between fuel pin centers) of ~2.6 cm. Figure 2 shows the conceptual design for the core configuration. Figure 3 shows an MCNP neutronic model for an 86 element core with a BeO reflector. The fuel element radial dimensions correspond to those typical of LWR fuel. The UO_2 density is assumed to be 10.3 g/cc. Both stainless steel and Zircaloy cladding material can be considered with Zircaloy selected as the baseline case. The selected pitch results from Monte Carlo analyses as the optimum spacing, for the given pin radial dimensions and fuel density, to minimize fuel requirements, while still assuring a robust negative water temperature/void reactivity feedback coefficient. Rapid fuel temperature negative reactivity feedback is provided through Doppler effects in the fuel and spectral and density effects in the moderator.

Numerous MCNP calculations have been performed to optimize the reactor design configuration. An adequate number of fuel pins must be present to allow for a critical condition at the operating temperatures of the fuel and moderator coolant. Additional reactivity must be available to overcome fission product poisoning and burnup. Different reflectors can be used to minimize the number of fuel elements required and the non-uniformity of the radial power distribution. Reflector options include water, graphite (C), beryllium (Be), beryllium oxide (BeO), and nickel (Ni). The most likely candidates include graphite, Be, or BeO. The fuel element pitch can also be optimized. A larger pitch can allow for a smaller number of fuel elements in the core; a smaller pitch can allow for a more negative moderator reactivity feedback but additional fuel elements.

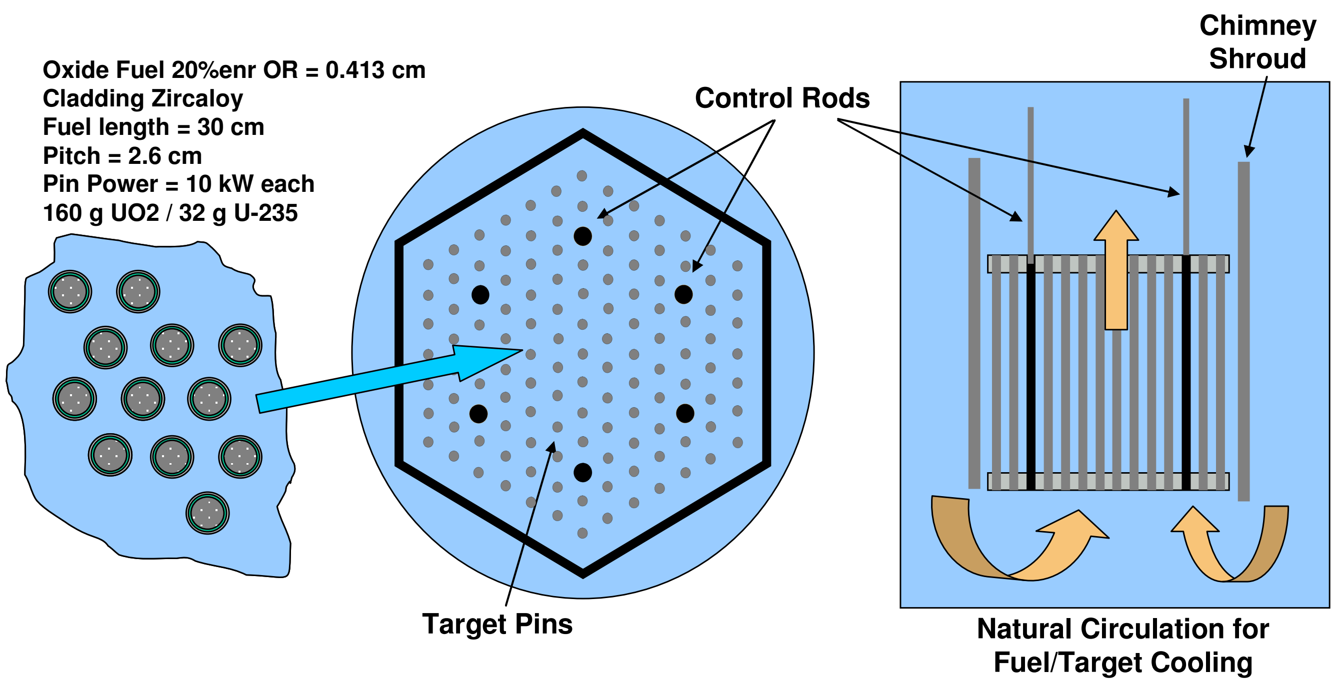

Figure 2. Fuel/Target Pins in a Triangular Lattice.

Figure 3 shows an MCNP neutronic model of a core configuration with 86 fuel elements 30 cm in length. The core is surrounded by BeO reflector elements. This core represents one of the smallest in size and number of fuel elements. The fuel region has a diameter of ~26 cm (10 in.). For this core the center region is a void that can be used for a safety element. Control elements can be placed in the reflector region around the core. The control and safety elements are moveable (up and down) rods loaded with a neutron absorber (poison) such as boron carbide (B_4C). The core maintains a large negative reactivity temperature coefficient and enough excess reactivity to operate at full power conditions.

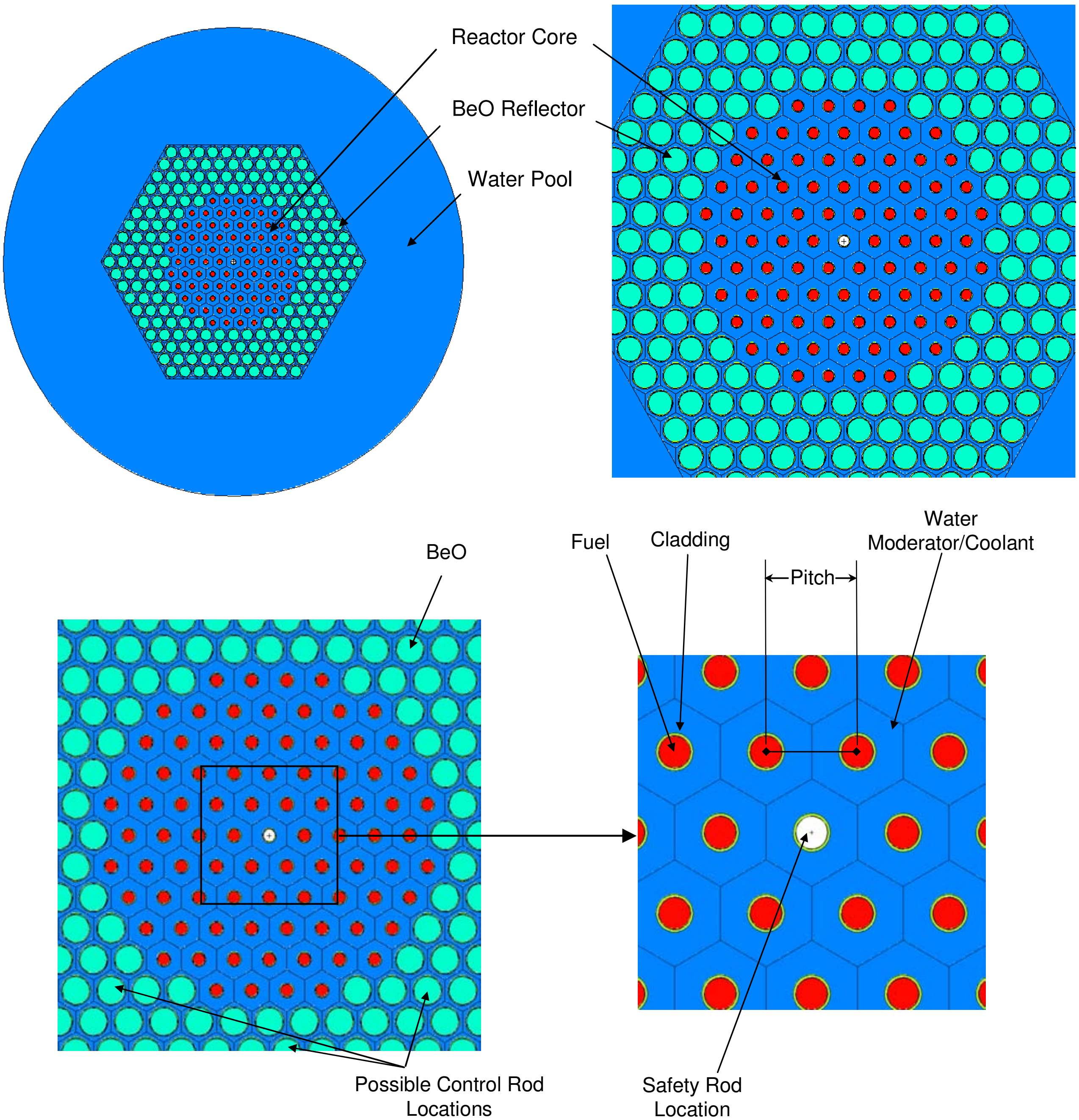

Figure 3. MCNP Neutronic Model of an 84 Fuel Element Core With a BeO Reflector.

An alternative approach is to replace the inner three target rows with a central un-fueled region constructed of a neutron-absorbing material such as stainless steel. An MCNP model is shown in Figure 4. The central cavity region provides an effective area for safety or control rods. The neutron absorbing liner serves to “flatten” the radial power profile across the fuel/target rows. Flattening the profile minimizes the power variation in the fuel/target pins and precludes the need to shuffle the pins in order to obtain a consistent power history during the irradiation cycle. A cut-away view is also shown in Figure 4. Possible locations for control and safety elements are shown.

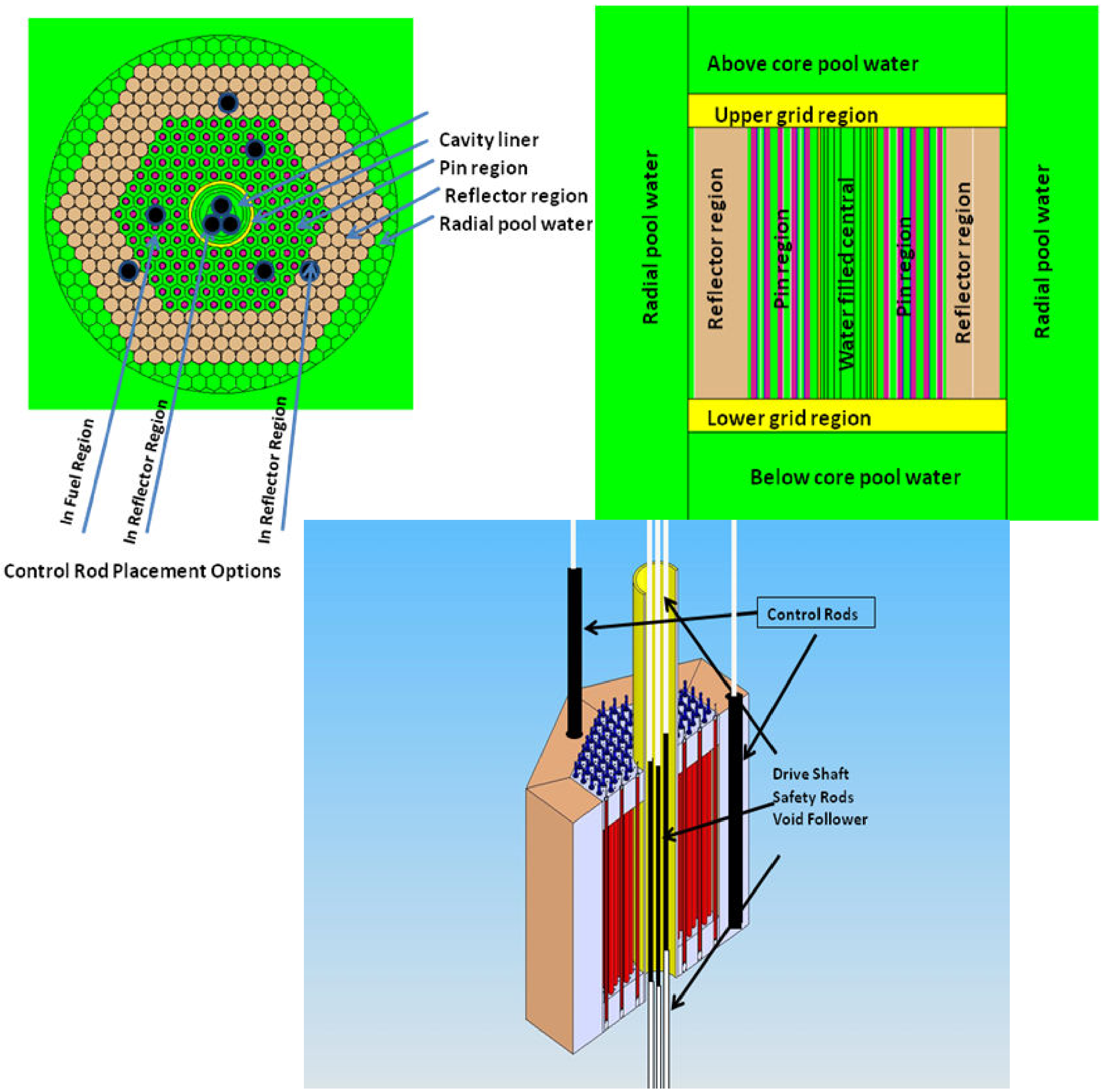

Figure 4. MCNP Neutronic Model of an Alternative Core With a Cavity Liner.

3.2 Fuel Pin Cooling

The simplest approach for cooling the reactor fuel pins is by natural-circulation flow through the core. As the water is heated in the channels between the fuel pins, the water density decreases, inducing flow through the core (bottom to top) without the use of pumps. This core cooling method is used in the ACRR and other university TRIGA type reactors.

Natural-circulation flow for an open pool was examined as a function of fuel pin power and compared with the limits of coolability, namely the critical heat flux (CHF). Results from preliminary analyses show that for an inlet temperature of 40 C in a 30 ft water pool, the fuel pin CHF is ~250 W/cm^2 (24 kW per fuel pin) without a chimney, and ~400 W/cm^2 (38 kW per fuel pin) with a hydraulic chimney to enhance natural-circulation flow. The limitation on peak fuel pin power will likely be due to power drop-outs and oscillatory behavior as significant subcooled boiling begins, causing chug flow in the coolant channels. Further analysis and experimentation are required to clearly establish the upper limit on the maximum power in which the fuel pins can operate. It is currently expected that 30 cm fuel pins should be capable of being operated in excess of 10 kW each (~100 W/ cm^2), with a maximum centerline fuel temperature of about 1200 C.

3.3 Performance

Conservatively assuming an average pin power of 10 kW, the reactor power would be 1.25 MW for a 125 element core, well in excess of the required target power to satisfy the USD for ^{99}Mo. To meet 100% USD with two days allowed for processing and shipping, ~16 target pins would be extracted and processed daily. Operating at higher pin power levels would reduce the number of pins processed per day and, concurrently, the total waste produced.

Operation at 20 kW of average pin power would yield greater than 200% USD. Subsequent analysis and evaluation of the chimney enhancement of natural-circulation flow will lead to higher limits on fuel pin power than the 10 kW assumed in the baseline concept. Even higher power limits can be achieved by use of cooling fins on the pins. The pin power is ultimately limited by the heat flux from the pin and/or the centerline temperature of the fuel pellet.

The market share for a start-up ^{99}Mo production operation will vary and be subject to available world supply. It is anticipated that during the initial start-up of the facility, the production operation would be expected to supply only a fraction of the USD, on the order of 20%. The system described above has the flexibility to operate to meet any portion of USD up to, and exceeding, 100% USD.

For any production level short of 100% USD, there are more irradiated targets present in the core than are required to be processed. For production of 30% USD, 38 fuel pins are required to be processed per week at a fuel pin power level of 10 kW. The remaining fuel pins could be designated driver pins and/or processed after longer irradiation times to optimize the production of long-lived non-^{99}Mo isotopes for medical or industrial applications. The pins can also be cycled on a period of longer than seven days, as long as the product specifications for purity can be met. A 21-day irradiation cycle may be possible, allowing for more of the fuel pins to be processed in the core.

Some examples of other isotopes produced by fission that could be extracted for medical or industrial applications include ^{131}I, ^{133}Xe, $^{140}Ba/^{140}La$, ^{141}Ce, ^{144}Ce, ^{147}Nd, $^{95}Zr/^{95}Nb$, $^{103}Ru/^{103m}Rh$, ^{105}Rh ^{153}Sm, ^{89}Sr, ^{91}Y and ^{147}Pm. Some fission product isotopes, such as ^{147}Pm used in micro-batteries, would be better suited for long-term pin irradiations, since they have longer half-lives. Activation production of other useful isotopes could be accomplished using pin locations outside the core grid locations.

3.4 Reactor Safety Features

The simplicity of the non-pressurized pool-type reactor system lends itself to inherent safety. Passive structures and the water pool are the main barriers providing protection for the workers and the public from accidents involving fuel pin cladding failures. Fission product source term buildup is low, due to the low power operation of the reactor and the removal and processing of the pins on a regular basis. Operating to supply 100% USD implies that all fuel is processed after 7 to 14 days of irradiation, which results in a very low core fission product inventory source term.

The cooling requirements for the low steady-state power level, and thus individual pin power, are adequately satisfied by inherent natural-circulation cooling. Forced cooling of the core is not required. Failure of the primary pool pumps, mechanical or electrical, is not an issue for the reactor. The reaction time required for addressing pool cooling system problems is quite long (tens of minutes) due to the heat capacity of the pool water.

Decay heat generation in the core after shutdown is also low. An emergency core cooling system or an emergency backup power system is not needed. Backup power for instrumentation is desirable, but not necessary. The reactor shuts down automatically with loss of electrical power. The electromagnetically-coupled control and safety rods drop into the core after a loss of electrical power, adding negative reactivity and shutting down the reactor. Since the core is cooled by natural-convection cooling, decay heat removal is also accomplished through natural-convection cooling to the reactor pool, which again has a very large heat capacity.

The temperature, void, and power reactivity coefficients for the reactor are strongly negative, giving the core the feature of self-regulation of power and inherent shutdown in over-power or over-temperature unplanned events.

The safety protection/control system can be simple and straightforward, directed at controlling reactivity and power level by adjusting the control rod positions. Control console instrumentation would be used for monitoring the neutron/gamma flux (reactor power), pool temperature, and radiation sensors in the high bay. The console and protection systems would be typical of ACRR or other university reactors. A TRIGA based console without the pulse capability would be sufficient for operating the reactor.

3.5 Fuel Pin Fabrication

Fuel pin/target fabrication is accomplished by simply using LWR fuel fabrication technology to produce LEU fuel/target pins of the desired enrichment and length. The ends of the pins would be capped with special end fittings for positioning and handling. Enhancement of cooling by adding cooling fins to the cylindrical pins could increase the heat transfer performance, but would also add to the fabrication cost. Trade and cost studies are in order to optimize the fuel performance and minimize cost.

The options exist to either buy the fully fabricated fuel pins from a LWR fuel vendor or a specialty manufacturer, or fabricate the pins on-site or locally. Again, trade and cost studies for operation and cost effectiveness are in order to determine the most appropriate fabrication scheme.

3.6 Fuel Pin Testing

The analyses, design, and limitations of the fuel pins can be validated using existing SNL facilities. Critical loading and zero power experiments can be performed in the currently operating SNL Critical Experiment Facility at TA-V.

Fuel/target pin power limits can be verified with out-of-pile (electrical heating) thermal-hydraulic experiments in any number of SNL facilities. Fuel pin testing to the desired power level using actual pins can be performed in the ACRR central cavity or pool.

The Annular Core Research Reactor pool could accommodate an actual full-power test program on the full-scale reactor system, if required.

3.7 Reactor Design Basis

The SNL Medical Isotope Reactor concept uses the ACRR as its design basis. The ACRR tank and reactor core are shown in Figures 5 to 7. The ACRR was originally designed by SNL staff and has been operated by SNL for over 30 years. The ACRR is an open pool-type reactor that uses natural-convection cooling of the fuel elements. The pool is cooled by a secondary cooling system. The fuel elements are 3.74 cm in diameter and 52 cm in length. The fuel is UO_2-BeO enriched to 35%. The core is made up of 236 fuel elements. A 9-inch diameter dry cavity extends from the center region of the core to above the pool to allow accommodation of a variety of experiments. It can operate in the pulse mode up to 300 MJ or steady-state mode up to 4 MW. A separate external cavity with TRIGA type fuel elements surrounding it is attached to one side of the ACRR to allow for larger experiments.

SNL has had extensive experience in not only design and operation of the ACRR, but in maintaining its safety documentation and training program and fielding complicated experiments (Ref. 22-24). Modifications to the ACRR have also been made for the previous medical isotope program and for other programs (Ref. 15, 17). Both the reactor control system and rod drive system were also designed and implemented by SNL staff.

The basic design and operational approach for the SNL Medical Isotope Reactor utilizes many of the same principles designed into the ACRR. It is this design and experience base with the ACRR and the previous medical isotope program which allows for a high degree of confidence that the SNL Medical Isotope Reactor will be successful. There are no technical or licensing issues that would impede the construction and operation of the reactor. The reactor can operate with a high degree of performance reliability and be cost effective.

Figure 5. ACRR Tank at Ground Level.

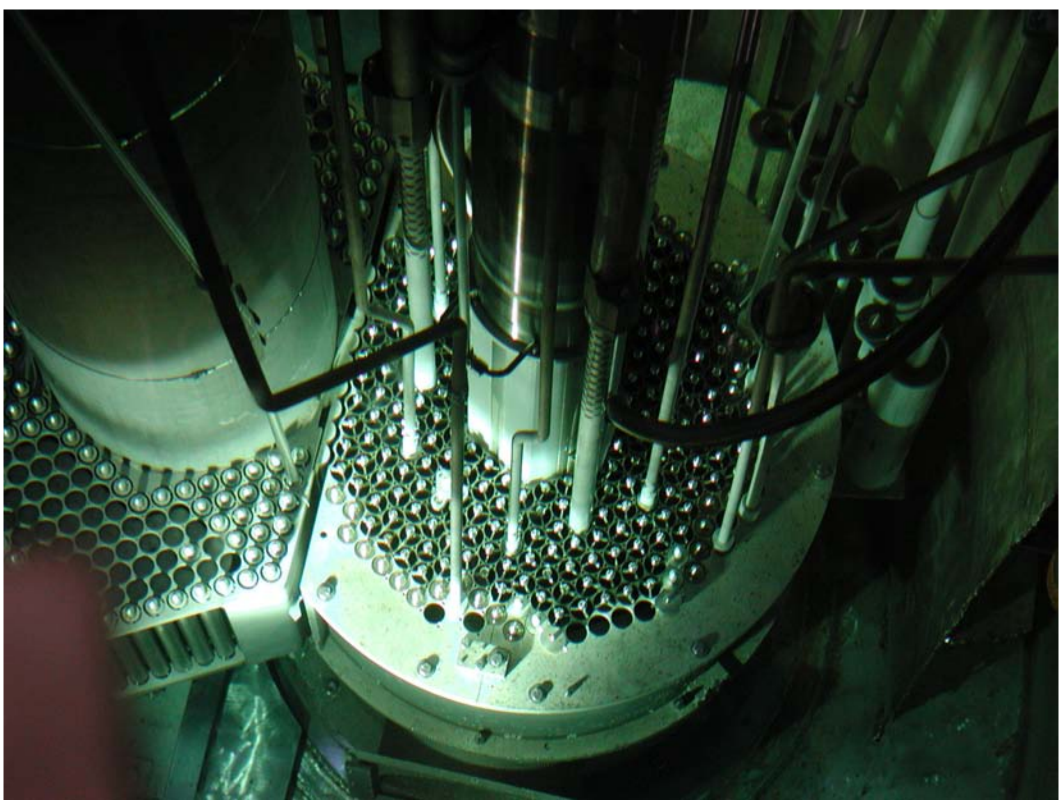

Figure 6. ACRR Core and External Cavity.

Figure 7. ACRR Core and External Cavity Operating at Power.

4 PROCESSING

Processing fuel pins to extract the ^{99}Mo would be performed on a regular schedule dependent on the customer's requirements. Since the half-life for ^{99}Mo is 65.9 hours (2.75 days) and the customer is billed for a six-day curie, it is important to minimize the time between removal of the target pin from the reactor and receipt of the ^{99}Mo product at the customer's facility in order to maximize the profitability of the facility. Approximately 1% of the ^{99}Mo product decays every hour. Typically, the time required to process and ship the ^{99}Mo product to the customer is one to two days. For a two-day processing and shipping time span, 40% of the ^{99}Mo product has decayed from the original quantity produced in the reactor. For a one-day time span, 22% of the ^{99}Mo product has decayed.

In order for processing to be efficiently performed, the reactor facility and hot cell processing facility must be collocated, and preferably attached to each other, with minimal transfer constraints. When the target/fuel pins are removed from the reactor core for processing, the pins contain tens of thousands of curies of fission products. Sufficient shielding of the pins must be provided in the transport to the hot cell processing facility to ensure acceptable radiation levels. The SNL design utilizes a simple transfer pool channel that is deep enough to allow for radiation shielding while maintaining a simple transfer approach.

As discussed previously, SNL was the last U.S. site to exercise the Cintichem process on a full-scale target for a full production cycle of seven days as part of the DOE-sponsored ^{99}Mo medical isotope program in the 1990's. The Cintichem hardware and SNL hot cell facility are shown in Figures 8 and 9, respectively. Unique processing hardware (not shown in the figure) was designed, fabricated, and tested to facilitate the Cintichem process requirements. The processing steps were proceduralized and evaluated for further improvement. Quality control procedures were followed to evaluate the product purity and concentration. ^{99}Mo product was shipped to a radio-pharmaceutical company for evaluation. The product was found to meet the required specifications for the drug master file (DMS). Radioactive waste was solidified and processed and later shipped to a disposal site.

With the experience and lessons learned from the previous isotope production program, SNL is in a unique position to continue to develop the Cintichem process on a large production scale. SNL recommends some effort be placed in automating much of the processing steps to minimize the burden placed on the hot cell manipulator operators.

As part of a new ^{99}Mo medical isotope program, SNL would use the Cintichem process, previously developed procedures, existing hardware, and existing expertise at SNL to engineer automated and easily maintainable hardware for target pin dissolution, extraction, purification, QC, and waste processing.

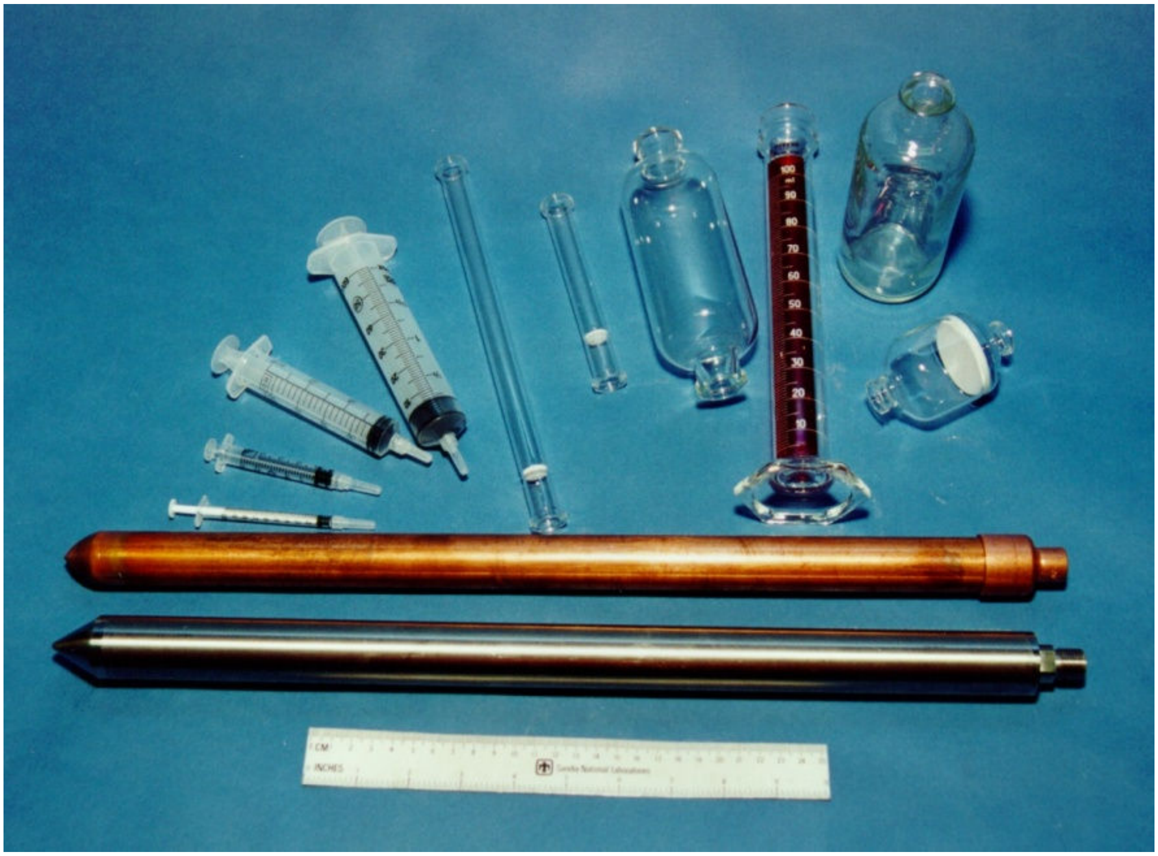

Figure 8. Cintichem Target and Processing Hardware.

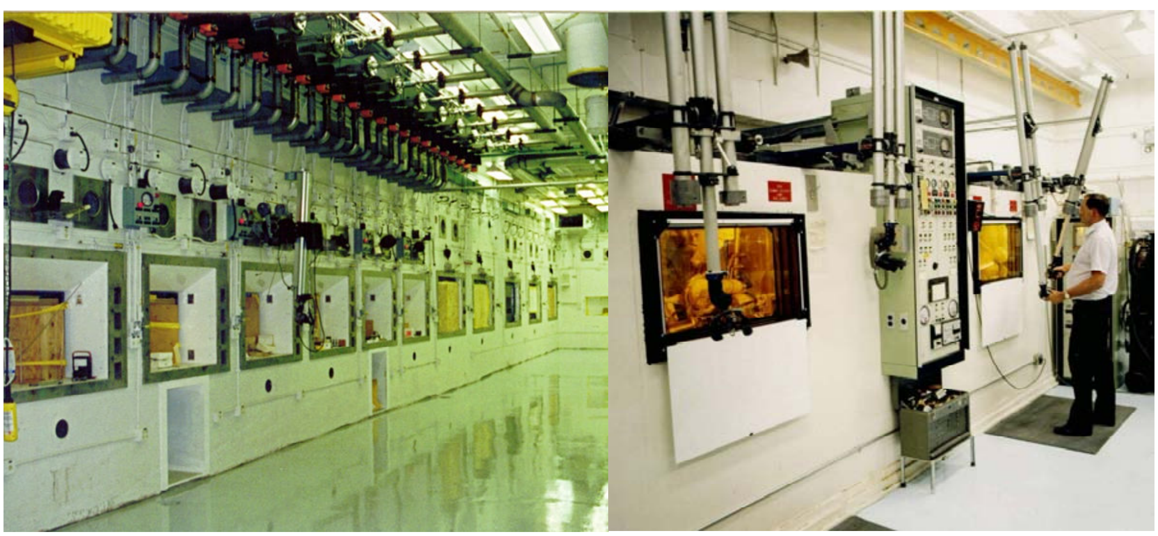

Figure 9. SNL Hot Cell Facility Modified for ^{99}Mo Production.

4.1 Processing Steps

The steps involved in processing a target/fuel pin are as follows:

1. After the required operating cycle, the reactor is shut down. The control system is used to place the reactor in the subcritical condition using the control rods.
2. The specified number of target/fuel pins is removed from the core. A special handing tool is used to remove the target/fuel pins from the core grid and place them in a storage basket in the transfer channel.
3. The removed fuel pin locations in the core are loaded with fresh target/fuel pins.
4. The reactor is restarted and restored to full power operation.
5. The irradiated target/fuel pins are moved to the hot cell transfer station. The basket holding irradiated fuel pins is moved under water in the transfer channel pool from the reactor facility to the collocated hot cell facility. The first cell in the hot cell facility is a transfer station that allows for removal of the target/fuel pins from the transfer channel into the shielded cell.
6. The irradiated target/fuel pins are moved from the transfer station to the dissolution cell for initial separation of the ^{99}Mo.
7. The irradiated target/fuel pins are opened and the fuel pellets removed and processed using the Cintichem or other extraction process.
8. Separated ^{99}Mo solution is moved to a purification cell. All other materials are moved to the waste processing and handling cell.
9. The ^{99}Mo solution is purified and prepared for shipping. A QC sample is drawn from the product.
10. The QC sample is analyzed.
11. The ^{99}Mo product is placed in a Department of Transportation (DOT) shipping container and sealed. The QC results and the shipping manifest are prepared.
12. The ^{99}Mo product shipping container is picked up by the freight transportation company and shipped to the customer.
13. The liquid waste is solidified and loaded with cladding and glassware into a waste drum.
14. The waste drum is stored in a shielded storage area until filled with the required number of waste hardware solids.
15. The waste drum is sealed when filled and stored for decay in a shielded storage area.
16. After a decay period (6 months to 1 year), the waste drum is sent to the disposal site in a DOT shipping cask.

The time required to process a fuel pin and have the ^{99}Mo product ready for shipping is several hours. The process timing must be such that the customer receives the ^{99}Mo product shipments on a regularly scheduled daily or weekly basis, set up by a contractual agreement. The production facility must optimize the conditions for processing to maximize their profits.

4.2 LEU Fuel

An integral part of the concept presented in this report is the use of LEU fuel in the reactor. Although the Cintichem process was developed for HEU fuel, there are no reasons why the efficiency or effectiveness of the process should be altered by using LEU fuel (Ref. 2). Using 20% enriched fuel, there will be about five times more uranium that will need to be dissolved in the dissolution step of the process. The volume of the dissolution cocktail for a 30 g uranium target using the Cintichem process is about 100 ml. For the current target/fuel pin conceptual design, approximately 160 g of uranium would be required to be dissolved, equal to about 500 ml of dissolution cocktail. This amount generated volumetric waste would still be very manageable. The volume of the dissolution cocktail will be dependent on the solubility of the uranium compound in the solution.

4.3 Waste Generation and Uranium Recovery Option

The waste generated for a target/fuel pin would include the fuel cladding, glassware, filter columns and liquid waste with the dissolved uranium fuel. The liquid waste would be solidified into a concrete form contained within a small steel vessel. The volume of waste per target would be expected to be on the order of two gallons (~8 liters) or ~0.25 ft^3. The waste would be placed into a 55 gallon drum. At least 25 targets, as waste, would fit within one 55 gallon drum. When the drum was filled, it would be sealed and stored in a shielded storage area for later disposition. The disposition path options have yet to be resolved.

The target/fuel pin processing described assumes that the liquid waste products are solidified and stored for later disposition at a disposal site. No reprocessing is assumed. However, depending on the cost recovery and added value, the uranium could be separated in the waste stream, purified, and sent to the fuel fabrication facility for reuse. This option will need to be studied further to determine the viability of reprocessing. Currently, none of the suppliers of ^{99}Mo reprocess the HEU, but instead dispose of it as waste. Viability of recovering the LEU includes issues such as cost effectiveness, product specification purity using reprocessed fuel, and fuel fabrication acceptance of reprocessed fuel.

4.4 Product Specifications

Two of the most important ^{99}Mo product specifications include the ^{99}Mo specific activity (^{99}Mo activity/total mass Mo) and the alpha emitter specific activity. The ^{99}Mo specific activity varies with the irradiation time of the fuel pin/target. The ^{99}Mo specific activity is 10,000 Ci/g Mo for a 7-day irradiation cycle, with 8-day decay (2 days for processing and shipping and 6-day decay period), 6,000 Ci/g for a 14-day irradiation cycle, and 4,000 Ci/g for a 20-day irradiation cycle. The requirement is that the specific activity exceed 1,000 to 5,000 Ci/g Mo for a 6-day Ci.

There are approximately 50 times more alpha emitting actinides generated using LEU fuel compared to HEU fuel due to the U-238 in the fuel. This should not, however, affect the product quality since the alpha emitting isotope that dominates this effect is U-234. It can be shown that the fuel would be required to be irradiated over 50 days before the alpha activity from actinide production would exceed the U-234 activity. The efficiency of the Cintichem process also maintains the alpha activity in the product to very low levels. The product specification for the alpha activity is 1×10^{-7} mCi-α/mCi-^{99}Mo.

Other ^{99}Mo product quality specifications include the gamma activity for ^{131}I, ^{103}Ru, ^{105}Rh, ^{132}Te, and ^{112}Pd, total gamma activity, and beta activity for ^{89}Sr and ^{90}Sr.

5 FACILITY DESIGN – REACTOR AND HOT CELL

The facility design layout for the SNL Medical Isotope Reactor concept is shown as an artist's conception in Figures 10 to 12 and a plan drawing in Figure 13. The facility design includes both the reactor facility and hot cell facility with attached office space. The overall facility would be sized with redundant features, where necessary, to ensure a highly reliable system. The hot cell facility would have redundant processing lines to allow for backup and capacity for greater than 100% of the USD for ^{99}Mo production. Process waste storage would be in a large shielded area at the end of the processing cell line. Waste storage capacity would be provided for at least one year at 100% of the USD.

Separate ventilation systems would be maintained for the hot cell facility and the reactor facility. A separation wall between the facilities maintains the ventilation separation between the facilities. The reactor does not require containment, since the fission product inventory is low and the large volume water pool acts to retain any volatile fission products that would be released in an accident. Each ventilation system would maintain filtered exhaust prior to release to the environment.

The reactor would be operated from an adjacent control room. Auxiliary rooms would include a storage area for fuel/target pins, chemicals, and miscellaneous equipment. Fume hoods for chemical preparation and QC would also be included in auxiliary rooms. Office space would be located outside of the reactor and hot cell facilities but within the same building structure and would maintain its own ventilation, heating, and cooling system. The facility would be required to meet the seismic requirements for the NRC at the site location.

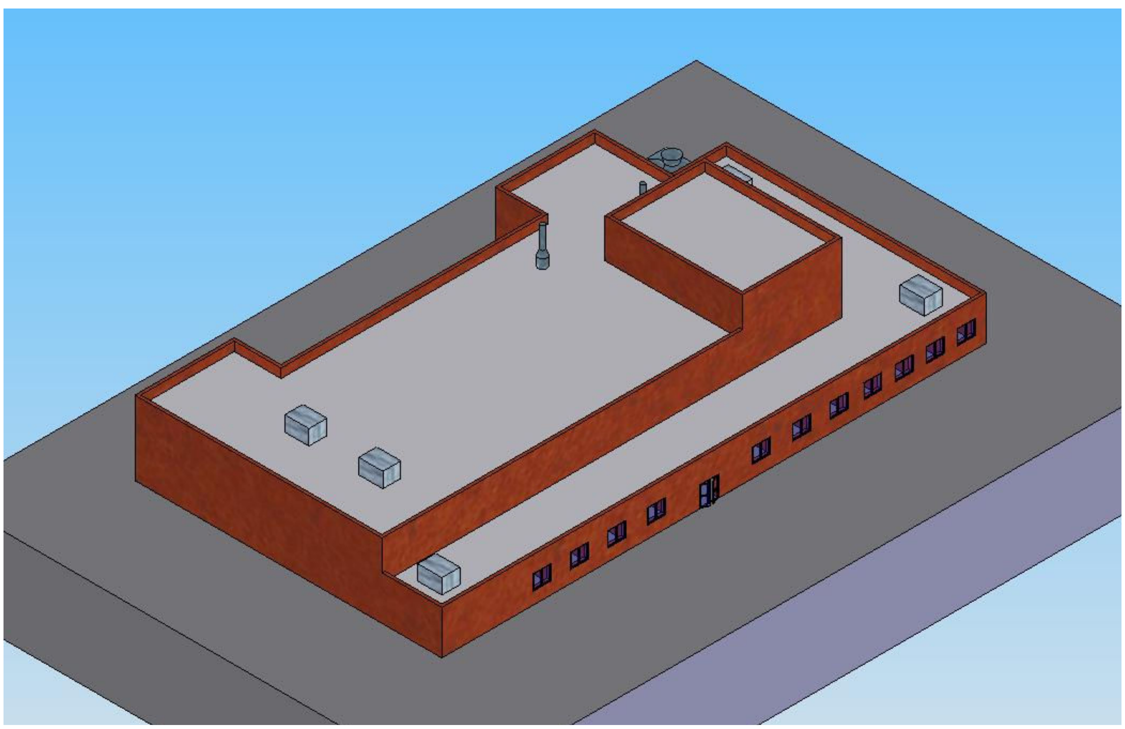

Figure 10. Aerial View of the ^{99}Mo Production Facility.

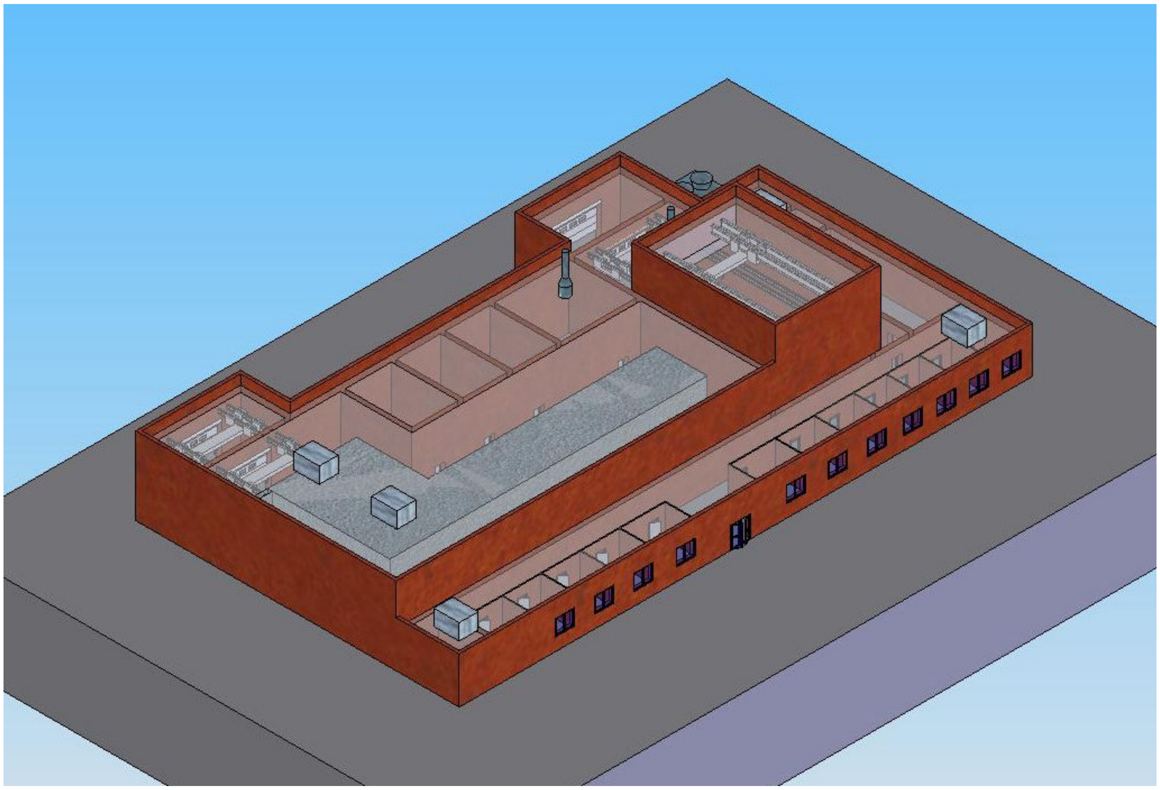

Figure 11. Production Facility With Roof Removed.

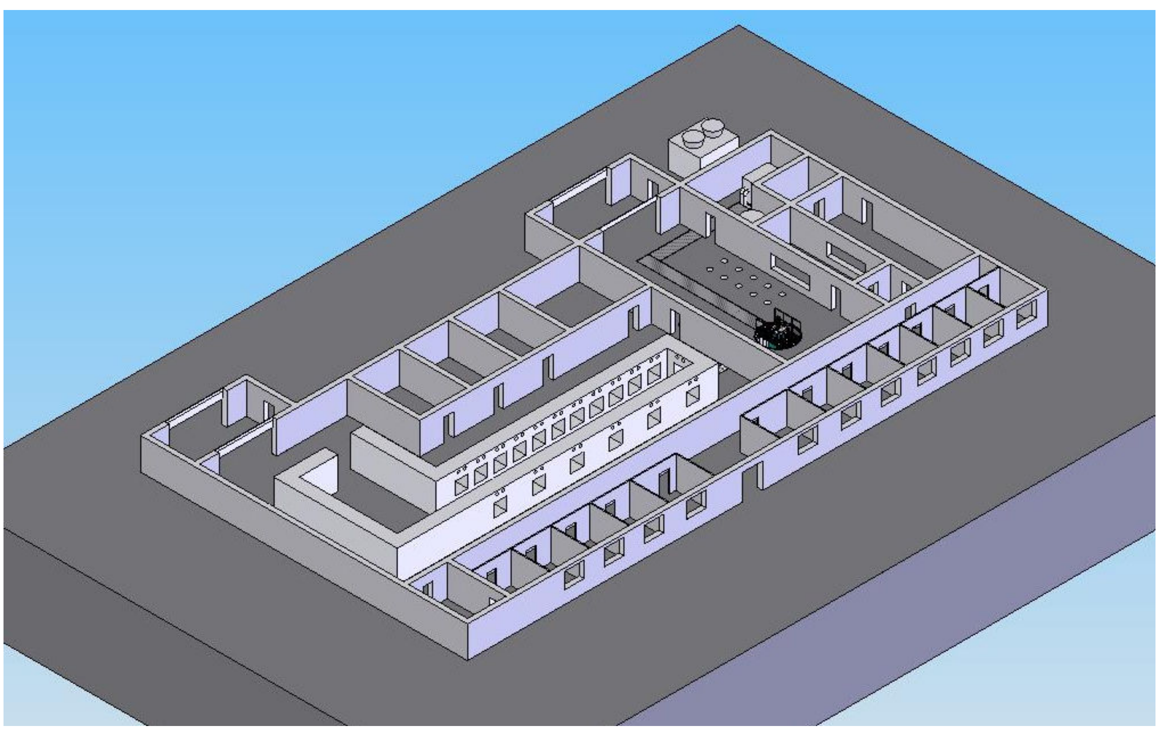

Figure 12. Production Facility Cutaway View.

The basic dimensions of the facility are shown in Figure 13. The facility footprint is approximately 20,000 square feet. An office area is shown in the right portion of the building and covers ~15% of the area. The reactor facility covers about 25% of the building area (top portion of the figure). The reactor area includes the high bay, equipment room, control room, and storage area. The hot cell facility covers the majority of the area and includes the hot cell processing lines, waste storage area, waste shipping area, chemical preparation labs, QC lab, and product shipping area.

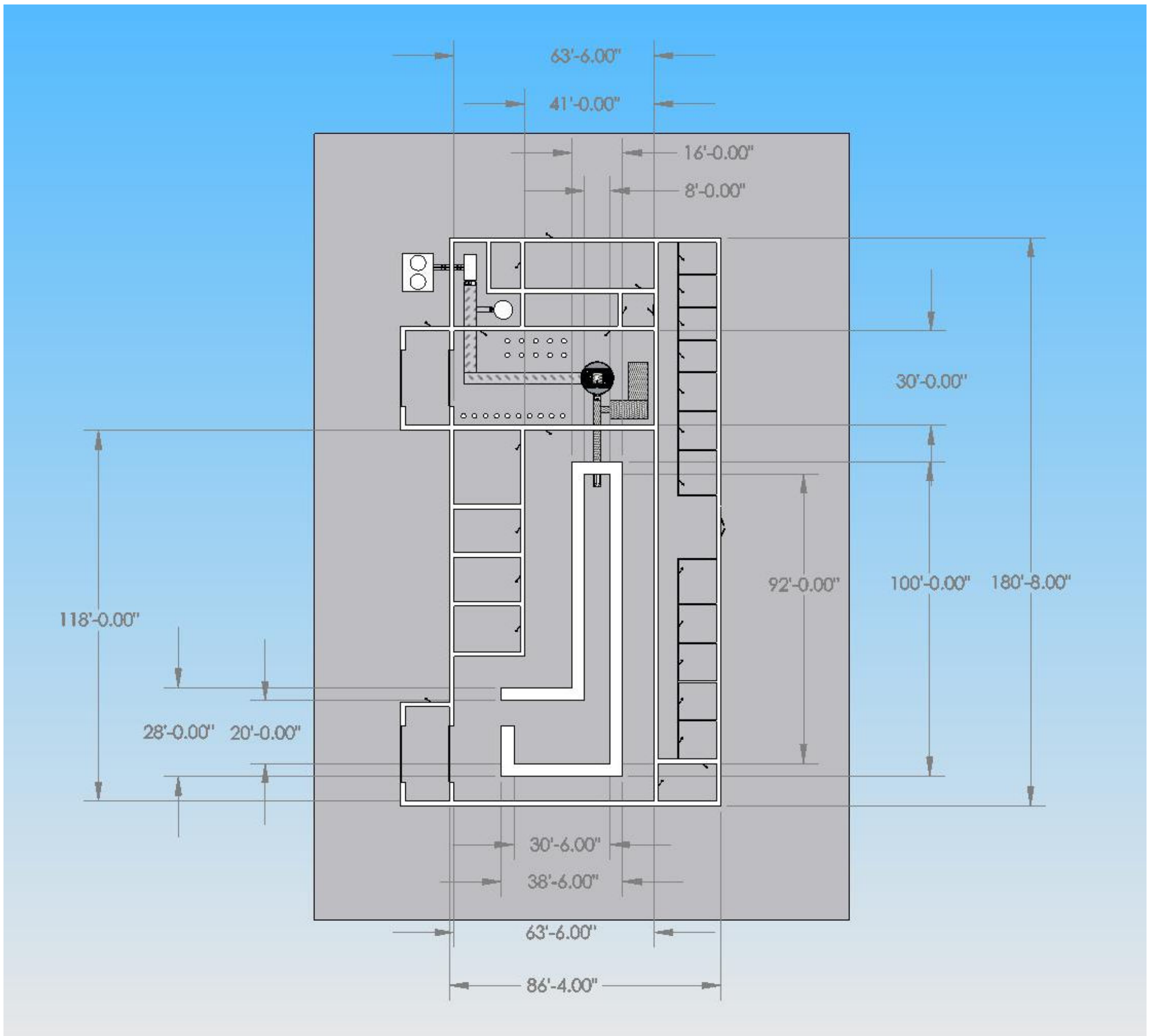

Figure 13. Production Facility Plan View With Dimensions.

5.1 Reactor Facility

The reactor high bay would have a footprint of ~30 ft by ~40 ft and ~30 ft in height. It would include a crane, ventilation system, makeup air supply, and a roll-up door for access to an external truck ramp. The walls of the reactor high bay would not be required to have shielding, since the transfer of irradiated pins would be performed under water. The reactor pool, transfer pool channel, storage pools, and storage pits would be located within the reactor high bay. The secondary heat exchanger and pump, as well as the water makeup and clean-up systems would be located within the equipment room next to the reactor high bay. Fresh fuel/target pin storage would be located within an auxiliary room next to the reactor high bay. The reactor would be operated from an adjacent control room.

The reactor and pool tank are shown in Figure 14. The tank would be cylindrical and ~30 ft deep. The tank would be constructed from stainless steel and would be ~10 ft in diameter. The top of the tank would be either at ground level or extend a few feet above ground level. The control system drives would be located on a bridge on the top of the pool tank. The water in the tank would be de-ionized and maintained using a clean-up system located in the equipment room. There would be no penetrations in the tank wall except for the transfer pool channel, which would be integrated with the tank but not extend to the bottom of the tank.

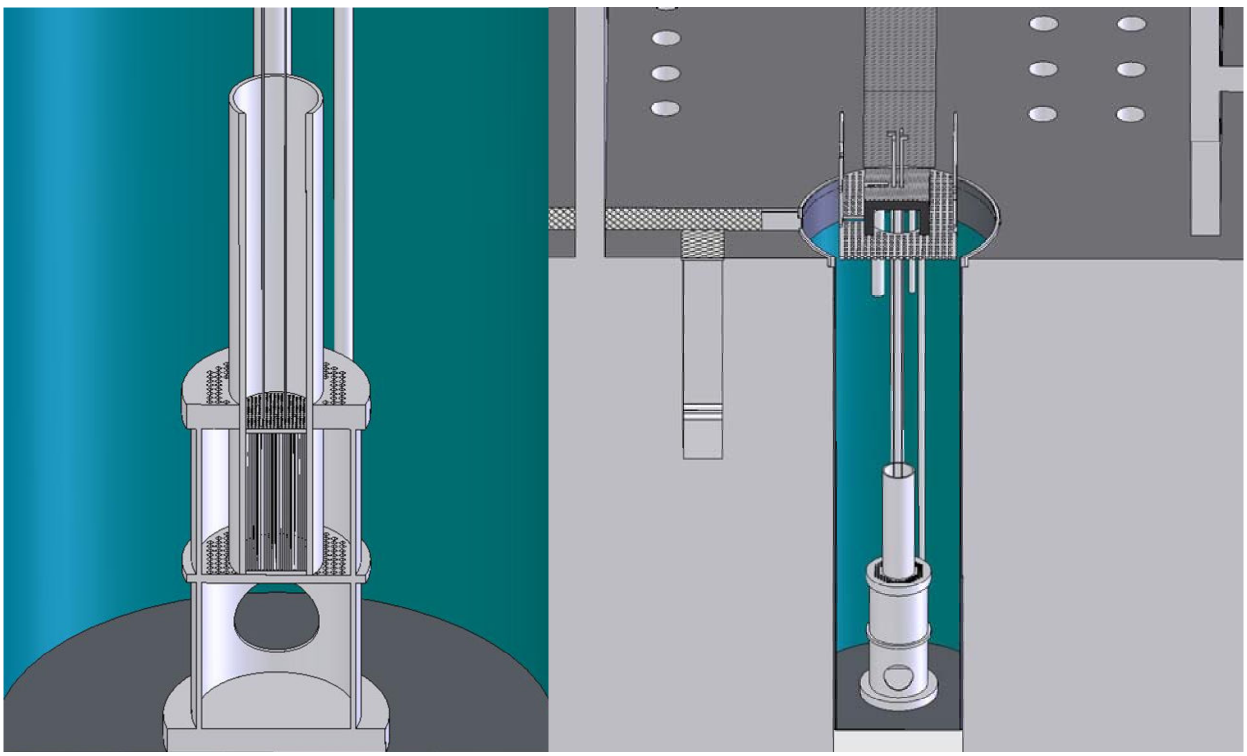

Figure 14. Reactor and Reactor Pool.

The pool water would be cooled by using a plate-type heat exchanger and pump located in the equipment room. Stainless steel piping would be used for the pool cooling outlet pipe that would extend to ~6 ft below the pool surface, as well as for the pool cooling inlet pipe that would extend to the bottom of the pool. The secondary water would flow through the secondary side of the heat exchanger through piping in the high bay wall, to a small 2 MW cooling tower located outside and adjacent to the building.

Irradiated target/fuel pin handling would be performed using automated or manual handling tools that would remove pins from the reactor grid with the reactor shut down and place them in a basket at the transfer channel. The empty locations in the reactor grid would be filled with fresh fuel pins in a similar manner. The transfer channel would be a stainless steel-lined trough ~1 ft wide and ~12 ft deep. It would extend from the pool tank to the first shielded cell of the hot cell facility. The irradiated target/fuel pin basket would be moved through the transfer channel to the first cell of the hot cell facility, where the irradiated pins would be removed from the basket using manipulators and prepared for processing.

5.2 Hot Cell Facility

The hot cell facility would be used for ^{99}Mo processing, purification, QC analysis, product packaging and shipping, waste processing, waste handling and storage, and waste shipping. The hot cell bay would incorporate redundant processing lines to allow for backup with capacity for greater than 100% of the USD for ^{99}Mo. The cells would be constructed with thick concrete walls for shielding and would include stainless steel processing boxes, shielded windows, manipulators, and ventilation filters. The processing boxes would include the necessary equipment to perform the task for that box, e.g., fuel removal and dissolution, product purification, waste handling and solidification, and product packaging.

Waste would be stored in the large shielded region at the end of the hot cell bay. Waste storage capacity would be provided for at least one year at 100% of the USD. Waste would be contained in 55 gallon drums and stored on a conveyer system. The drums would exit into a shielded room after the required decay period (at least 6 months) and shipped to a disposal site using a DOT shipping cask.

Ventilation would be controlled in three zone volumes in the hot cell facility. Zone 1 would be the ventilation control volume within the stainless steel processing boxes. Zone 2 would be the control volume within the shielded concrete structure (hot cell bay) and waste storage area. Zone 3 would be the control volume within the remaining region of the hot cell facility, including the area inhabited by the operating staff. Zone 2 would be maintained at a lower pressure than Zone 3, and Zone 1 at a lower pressure than Zone 2, to ensure that airflow would always be from the non-contaminated volume of Zone 3 to the potentially highest contaminated volume of Zone 1. Each of the zones would have its own filtration and ventilation system equipment. The exhaust from the facility would be routed through a stack located outside the facility. The ventilation system would be designed to minimize radioactive releases to the environment. The releasable quantities would be established through the NRC, Environmental Protection Agency (EPA), and local agencies.

5.3 Construction Cost and Time

The facility design layout is currently in a pre-conceptual stage. The construction costs and construction time have yet to be estimated. Recently constructed nuclear facilities of similar size and function can be used to estimate the cost and time of construction. The objective of the facility is to have adequate capacity to ensure that 100% of the USD for ^{99}Mo could be met in a cost-effective manner. The goal would be to maximize the functionality of the facility while minimizing the extraneous bells and whistles. Using this approach and past experience with processing at SNL, a robust facility design is achievable. It is estimated that this facility with a complete safety basis could be built for less than $100M and constructed within five years.

6 SITING

The facility concept presented in this report could be sited at a variety of locations due to the relatively low hazard classification and fission product inventory. It is desirable that the site be near a major airport to allow for minimal shipment times, but not essential if major airports are only a few hours away by ground transportation, or if cargo or chartered flights can be made through the local airport that can carry the DOT-approved, shielded-product shipping container. If the facility produces in excess of 20% of the USD for ^{99}Mo, radio-pharmaceutical companies would be encouraged to build and operate ^{99}Mo generator plants nearby. This would minimize the transportation requirements and lessen the timing demands for the process.

Since the reactor will only operate at a power level of 1 to 2 MW, and a significant fraction of the target/fuel pins will be replaced regularly with fresh pins, the fission product inventory, or source term, in the reactor core is relatively low. Since the reactor is operated in a large water pool, the release of fission products from accidents involving ruptured fuel pins will be limited to mostly noble gases, due to water scrubbing of the halogens and other volatile fission products. Downwind dose estimates to the public for both minor and major accidents involving the reactor will be low.

The collocated hot cell facility will have only a few target/fuel pins in process at one time. Hence, the source term for accidental release from the hot cell portion of the facility will be limited. As described in Chapter 4, the hot cell facility will have redundant zones and filtration systems to limit the releases to the public and environment under both normal and accident conditions.

7 LICENSING

It is anticipated that, regardless of the funding source for the facility, the construction and operation of the facility would be licensed under the authority of the NRC, and the facility would be constructed and operated as a private commercial endeavor. The role that the DOE and SNL would play in facilitating this endeavor is not certain. Depending on potential funding through the DOE, SNL could play a large role in advancing and developing both the reactor and hot cell processing technology.

The characteristics of the reactor facility fall within the operating envelope of existing university research reactors, which the NRC is currently charged with regulating and for which regulatory requirements and guidelines exist. The fact that this would be a commercial operation rather than a university research operation will require a more in-depth safety basis, quality assurance program, configuration management process, and training program to be established. The NRC did regulate the Cintichem operation in Tuxedo, New York, prior to its planned shutdown. Except for the different fuel form and fuel configuration, the proposed concept is similar to the Cintichem operation regulated by the NRC. No technology with which the NRC does not have copious experience is presented by this concept. The reactor is typical of university research reactors and the fuel adapted from commercial LWR fuel systems with which the NRC has much experience.

The reactor concept is a low steady-state power system with low fission product inventory (i.e., low source term), robust negative feedback coefficients, passive cooling by natural-convection cooling, and passive shielding and radionuclide scrubbing by the large volume of pool water. The reactor system shuts down on loss of power and the only backup power needed is for lighting and monitoring instrumentation. The balance of plant equipment included in the water cooling system, purification system, and make-up water system is straight forward in design and operation. The control console that includes the reactor protection equipment and control instrumentation is also straight forward and is used at many research reactor facilities. The fuel/target pin fabrication utilizes proven technologies and fabrication processes based on many years of LWR history.

The hot cell facility and processing aspects of the operation are straightforward and use well-developed technologies for shielding, processing and ventilation. The radioactive source term is limited to the number of targets that are in the process of being handled at any given time. Packaging and shipping of product and waste must conform to established NRC and DOT regulations.

The preparation of safety basis documentation is straightforward and well within the capabilities and experience of the SNL staff. Both research reactors and hot cell facility safety bases have been previously prepared at SNL for facilities that would be similar to that presented for this concept. Except for the fuel materials and the fuel/target grid arrangement, the proposed reactor system is similar to the open pool-type reactor systems operated at universities around the country and regulated by NRC.

8 PRODUCTION CAPACITY AND COST EFFECTIVENESS

The USD for ^{99}Mo is ~6,000 six-day Ci per week. The world demand for ^{99}Mo, excluding the U.S., is also ~6,000 six-day Ci per week. The selling price for ^{99}Mo from the supplier is ~$470 per six-day Ci (Ref. 2). This means that the potential revenue stream for the production of 100% of the USD for ^{99}Mo is ~$2.8M per week, or ~$147M per year.

^{99}Mo is generated by the fission process at the rate of 0.061 atoms of ^{99}Mo per fission. This translates to a saturated activity of 51.4 Ci per kilowatt of target power. Using a half-life of 2.75 days (66 hours) for ^{99}Mo, a 7-day target irradiation will reach 82.9% of the saturated activity (42.6 Ci/kW); a 14-day target irradiation will reach 97.1% of the saturated activity (49.9 Ci/kW).

The ^{99}Mo activity in units of Ci/kW as a function of time is shown in Figure 15 for a 7-day target irradiation with a 2-day time period for processing and shipping and a 6-day decay period. The activity at the end of the 6-day decay period is what is known as the 6-day curie. It represents the quantity of ^{99}Mo that is invoiced to the radio-pharmaceutical customer. For this irradiation time and processing and shipping time, the number of 6-day curies produced is **5.66 Ci/kW**. Decreasing the processing and shipping time by one day will increase the ^{99}Mo activity by ~29%. Therefore it is crucial to minimize the processing and shipping time in order to maximize the profitability of the ^{99}Mo process.

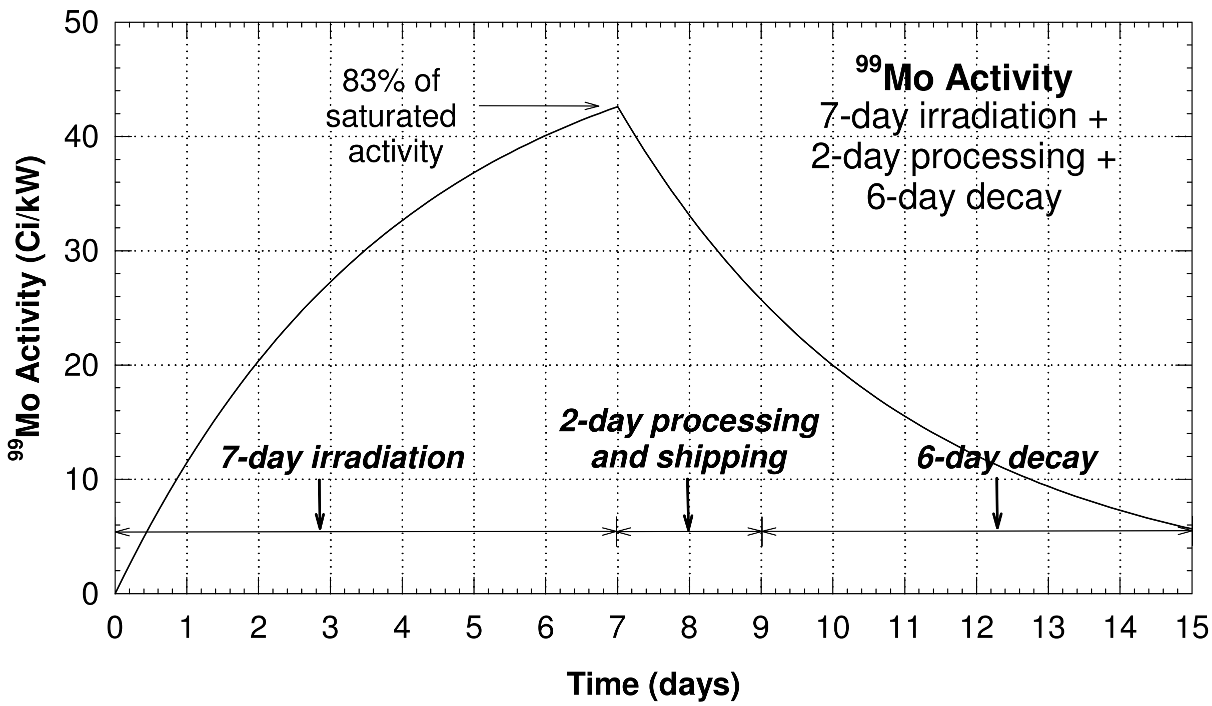

Figure 15. Activity (Ci) of ^{99}Mo per Kilowatt as a Function of Time for a 7-day Irradiation With a 2-day Processing and Shipping Period and a 6-day Decay Period.

The quantity of the ^{99}Mo produced in the target/fuel pin targets is directly proportional to the power generated in the pin. One uncertainty in the concept design is the maximum power at which a 30-cm-long target/fuel pin could operate. It is anticipated that the fuel pins could be operated to at least 10 kW of power and potentially higher. At a power level of 10 kW, ~21 fuel pins would need to be processed each week to meet 20% of the USD. Approximately 106 fuel pins would need to be processed each week to meet the full 100% of USD.

At a production level of 20% of the USD for ^{99}Mo, ~$0.56M of revenue would be generated each week, or $29.3M per year. The cost for targets, chemicals, glassware, and waste disposal would be ~$0.1M per week or ~$5.1M per year at a 20% production level. Assuming $5.6M per year for staff salaries, net profit could be ~$18.6M per year at a 20% production level. Increasing the production level above 20% would only increase the expenses incrementally. Assuming the capital cost for the facility of ~$100M, the break-even time would be ~5 years at a 20% production level.

With the selling price for ^{99}Mo at $470 per 6-day Ci, it is reasonable to assume that a facility using the target/fuel pin reactor concept presented in this document could be cost effective, at a minimum production level of 20%.

9 CONCLUSIONS

SNL has developed a medical isotope reactor concept that can meet and exceed the USD for ^{99}Mo. The concept developed is flexible in that it can either meet a few percent of the USD or exceed 100% of the USD. The key feature of the concept is that the fuel pins in the reactor are also the targets. There is no need for a high-power driver core in addition to target/fuel pins. The target/fuel pins are very simple in design, drawing from LWR technology and the use of LEU fuel enriched to less than 20%. The reactor operates at a relatively low power level of 1 to 2 MW, is small in size, and maintains negative reactivity coefficients and a low fission product inventory. It operates in a water pool, uses natural-convection cooling to cool the fuel pins, and is passively safe. The water pool is cooled by a secondary cooling system. No backup power supply or emergency core cooling system is required. Loss of electrical power shuts the reactor down by dropping the control and safety rods. The reactor is controlled by using a straightforward control system similar to that found in university-type research reactors. The target processing uses the well-developed Cintichem process. The fuel pins are removed from the reactor on a regular cycle and processed at a collocated hot cell facility.

There are no new technologies that must be developed to implement the SNL Medical Isotope Reactor concept. The concept can be made cost effective at a production level of 20% of the USD. There are no impediments that prevent this type of reactor, along with its collocated hot cell facility, from being designed, fabricated, and licensed today.

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